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CALCULATED POWER DISTRIBUTION OF A THERMIONIC, BERYLLIUM OXIDE REFLECTED, FAST-SPECTRUM REACTOR

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16. Abstract

A calculational procedure is developed and used to calculate the detailed power distribution in the fuel elements next to a beryllium oxide reflector of a fast-spectrum, thermionic reactor. The results of the calculations show that, although the average power density in these outer fuel elements is not far from the core average, the power density at the very edge of the fuel closest to the beryllium oxide is about 1.8 times the core average.

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SUMMARY

By using the calculational procedure described in this report, it is possible to calculate the detailed power distribution in the fuel elements of a fast-spectrum, thermionic reactor which are next to a beryllium oxide reflector. Since the neutron energy spectrum is rapidly varying in this core-reflector interface region, the calculational procedure includes a method for accurately representing the leakage of neutrons in the axial (or transverse) direction of a two-dimensional Rθ calculation, which has a fine enough energy group structure and spatial mesh to give an accurate power distribution. Not only the radial but the azimuthal power distribution in any particular fuel element can be calculated.

The results of the calculations show that, although the average power density in the outer ring of fuel elements is not far from the core average, the power density at the edge of the fuel next to the reflector is 1.95 times the core average. This is the value for the smeared fuel cell. With the 1 mm (0.040-in.) tungsten clad put into the calculation in a discrete manner the power density at the core edge is calculated to be about 1.8 times the core average. This high power density decreases rapidly on going toward the center of the fuel element, and it gets down to the core average value about 1 centimeter into the fuel.

INTRODUCTION

The problem of power peaking at the edge of a fast-spectrum reactor with a neutron moderating reflector has been known for some time. This problem exists because the nuclear fuel has a much larger fission cross section for the slower moving, moderated neutrons returning to the core from the reflector than it does for the fast neutrons in the core. Thus the returning moderated neutrons create a large increment of fissioning at

the edge of the core. This local peaking was experimentally measured in critical experiments performed with beryllium and beryllium oxide reflectors on the General Electric Company's 710 program in 1965 (refs. 1 and 2).

Calculations (such as those made for ref. 3) indicated this power peaking, but because of the large local variation of neutron spectrum it is necessary to have a sufficient number of energy groups and a sufficiently fine spatial mesh to get an accurate estimate of the local power distribution at the core edge. Calculations that were made for a reference thermionic reactor (ref. 4) did not have sufficient spatial and spectral detail to show this rapid increase in power density at the core edge.

In detailed two-dimensional analyses the mixed spectrum at the core edge makes questionable the usual analytical method of accounting for the transverse neutron leakage from a planar calculation by the assumption of an effective core height. A more accurate treatment of transverse leakage in two-dimensional transport calculations is presented. This method is applied to a three-dimensionally fuel-zoned, fast-spectrum, thermionic reactor which had a neutron moderating reflector of beryllium oxide. The purpose of this work was to get an accurate power distribution for ultimate comparison with the measured power of a critical experiment that was later performed on a mockup of a segment of this thermionic reactor (ref. 5).

CALCULATION METHOD

In this core design the amount of nuclear fuel in each of the diodes depends on its position in the core (ref. 6). Because the diode fuel loading is varied in both the radial and axial directions in the core, it presents a three-space dimension problem. This can be approximated by a cylindrical geometry solution (RZ) (as shown in fig. 1), but the cylindrical outer boundary of the core is not a good representation of the actual irregular boundary. (The diode fuel loadings are also shown in fig. 1 and will be discussed later.) A horizontal slice through a layer of diodes is amenable to treatment using Rθ geometry; however, a method of properly accounting for the transverse leakage into or out of the slice must be developed. The appendix describes the method developed which makes use of the RZ solution to obtain spatially dependent transverse leakage cross sections for use in the Rθ calculations. The Rθ geometry (fig. 2) permits a more realistic representation of the outer periphery of the core than does the RZ geometry. Several local-to-average power density ratios (fission density) are indicated in figure 2; these will be discussed later.

As pointed out in the appendix, the use of a single-space independent buckling or, equivalently, an effective height for reactors with mixed spectra between core and reflector can result in large discrepancies in the multiplication factor and power distributions. The method developed to treat transverse leakage accounts for both the three-

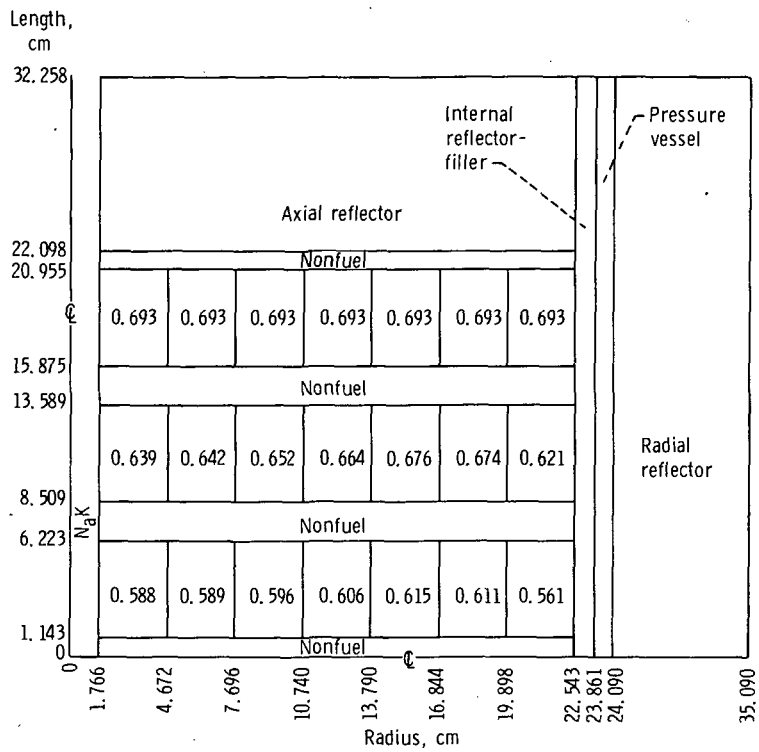

Figure 1. - RZ calculation model showing fuel zoning.

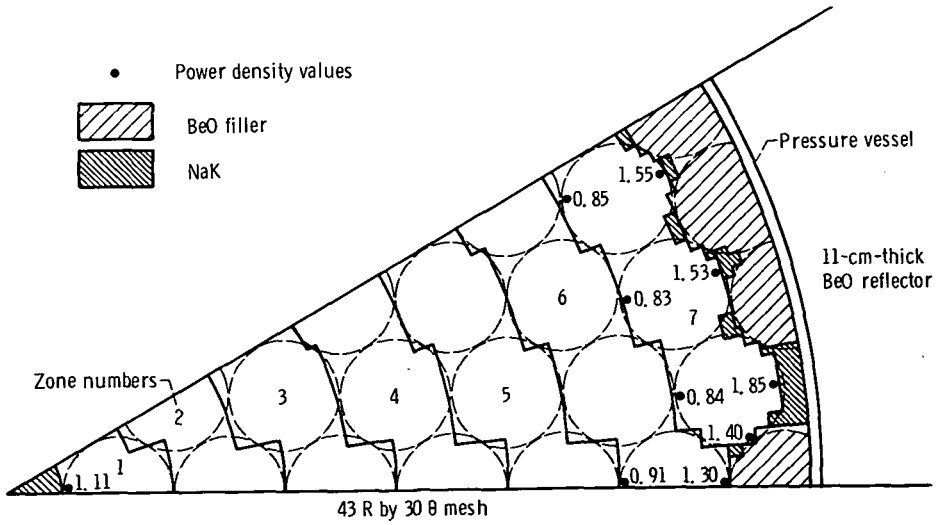

Figure 2. - RΘ calculation model for reference thermionic reactor.

dimensional fuel zoning and the fast core-moderating reflector mixed spectrum problems.

In the present analysis a 20-energy group P_1 set of cross sections was used. The 13 fast-group cross sections are from the GAM-II program (ref. 7), and the seven thermal groups ($E \leq 0.414$ eV) are from the GATHER program (ref. 8) using gas kernel calculations at 1000 K for the sodium potassium eutectic (NaK) coolant and beryllium oxide (BeO) reflectors. Spatial multigroup solutions were obtained using the DOT program (ref. 9). Both $R\theta$ and RZ geometric models were used with the S_4 discrete ordinate approximation. Material specifications for each region of the RZ model (fig. 1) are given in table I.

TABLE I. - COMPOSITIONS FOR RZ CALCULATION

[3.363-cm pitch.]

Region	Material	Volume fraction
Fueled	[a]UCZrC	[b]0.5465
	W	[c].098
	Nb	.222
	Al_2O_3	.021
	NaK	.093
Unfueled	Zr	0.0006
	W	.1324
	Mo	.0089
	Nb	.3146
	Al_2O_3	.0862
	NaK	.0931
	Ta	.0275
Filler	BeO	0.714
	Nb	.193
	NaK	.093
Vessel	Nb	1.0
Shutters	BeO	0.95
	Nb	.025
Axial	BeO	0.803
Reflector	Nb	0.098
	NaK	.093

[a]90UC-10ZrC(4W/DW), 0.72 theoretical density, 10 percent void.

[b]Core average fuel loading.

[c]Includes W from fuel.

RESULTS OF CALCULATIONS

RZ Calculation

The calculated power distribution for each fuel zone is shown in figure 3. The RZ geometric model was used assuming axial symmetry; this is a good assumption and reduces computer time significantly. The fuel fractions indicated in figure 3 are based on the volume occupied by the fuel relative to the volume of the emitter cavity. The fuel is made of 90 mole percent uranium carbide and 10 mole percent zirconium carbide. Then 4 weight percent tungsten is added. The uranium is enriched to 93.2 percent uranium-235. For this reactor design the maximum fuel fraction is 0.693 to allow for fuel swelling. As seen in the figure, the power distribution by zone is not very uniform, but some additional flattening could be done by moving fuel from the center zones to some of the outer zones that are low in relative power.

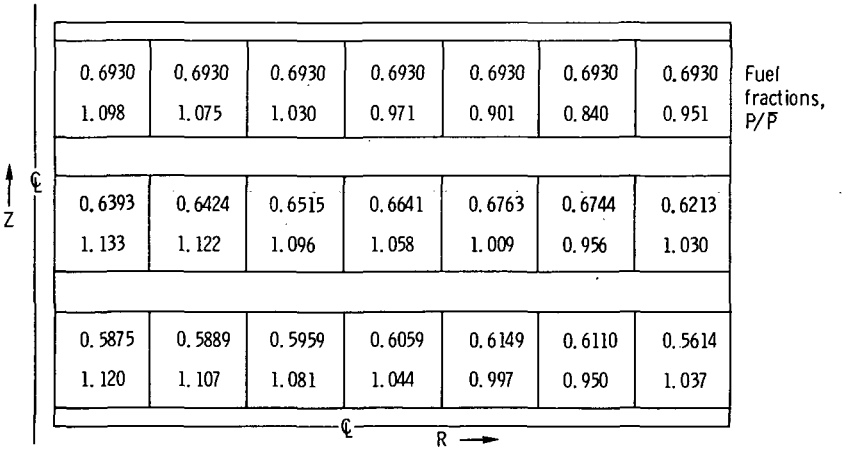

Figure 3. - Power density ratio $P/\bar{P}$ distributions by zone from symmetric RZ calculations.

Rθ Calculation

An $R\theta$ calculation for the center set of diodes was performed using the method is given in the appendix. This particular slice was chosen because the power peaking at the core-reflector interface would be greatest for this case. The geometry used is that shown in figure 2. Also shown in figure 2 are several local-to-average fission density ratios, which are mesh interval averages. Note, in figure 2, that there is a rapid decrease in the fission density ratio from the core edge inward. For example, in zone 7 for the pin nearest the reflector the decrease is from 1.85 to 0.84 across the diameter of the pin.

Figure 4 shows the fission density ratios for each fueled diode in the center slice as calculated with the Rθ model. Zone values are also given. The agreement between the Rθ and RZ results (fig. 3) is good except for zone 7 in which the RZ value is about the same as for the pin with the largest fission density ratio. This points out the necessity of using the Rθ (or possibly XY) geometry to adequately treat the irregular boundary of the core.

Figure 5 shows a number of fission density ratios for the diode nearest the pressure vessel. About 92 Rθ mesh cells were used to define this diode cell. Note the strong azimuthal dependence and the steep decrease in fission density from the edge nearest

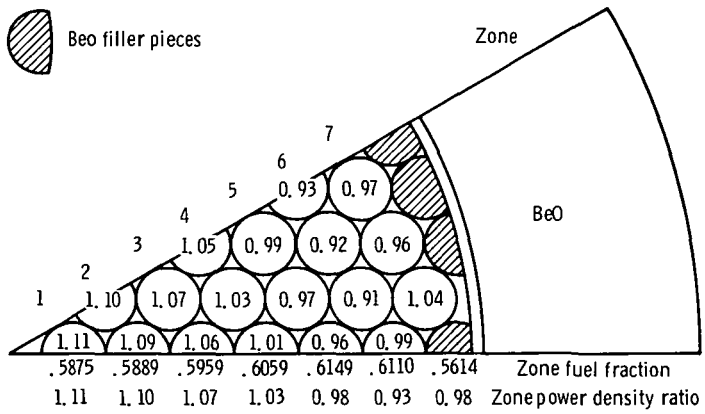

Figure 4. - Fuel pin and fuel zone power relative to average in Rθ calculations for center traverse. Multiplication factor, 1.057.

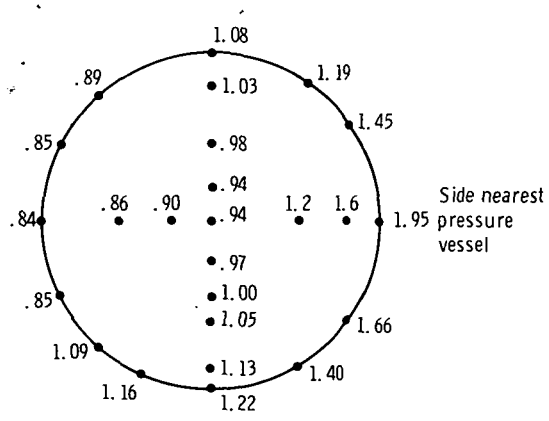

Figure 5. - Fuel cell relative power distribution at selected points. Cell average power density ratio, $P/\bar{P}$, 1.04; average power density, $\bar{P}$, 1.0 for Rθ calculation.

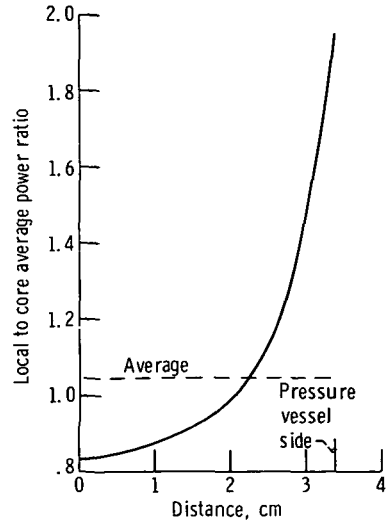

Figure 6. - Fuel pin cell nearest to pressure vessel. Fuel fraction. 0.5614.

the reflector in toward the center. This is more apparent in figure 6, which is a plot along the diode diameter. The fuel could be subjected to local swelling because of this local hot spot and cause swelling of the emitter. With only a 0.0254-centimeter clearance between emitter and collector, an electrical short could occur. Large temperature gradients would not be expected because of the excellent thermal conductivity of the materials. However, these calculations were not done and would need be done before the validity of these statements concerning fuel swelling and temperature distribution could be determined.

Figure 7 shows neutron energy spectra calculated from the Rθ results for various fuel zones. The fluxes are power normalized to 1.575 thermal megawatts which is the average power between the beginning and end of 20 000 hours of life. The spectra for

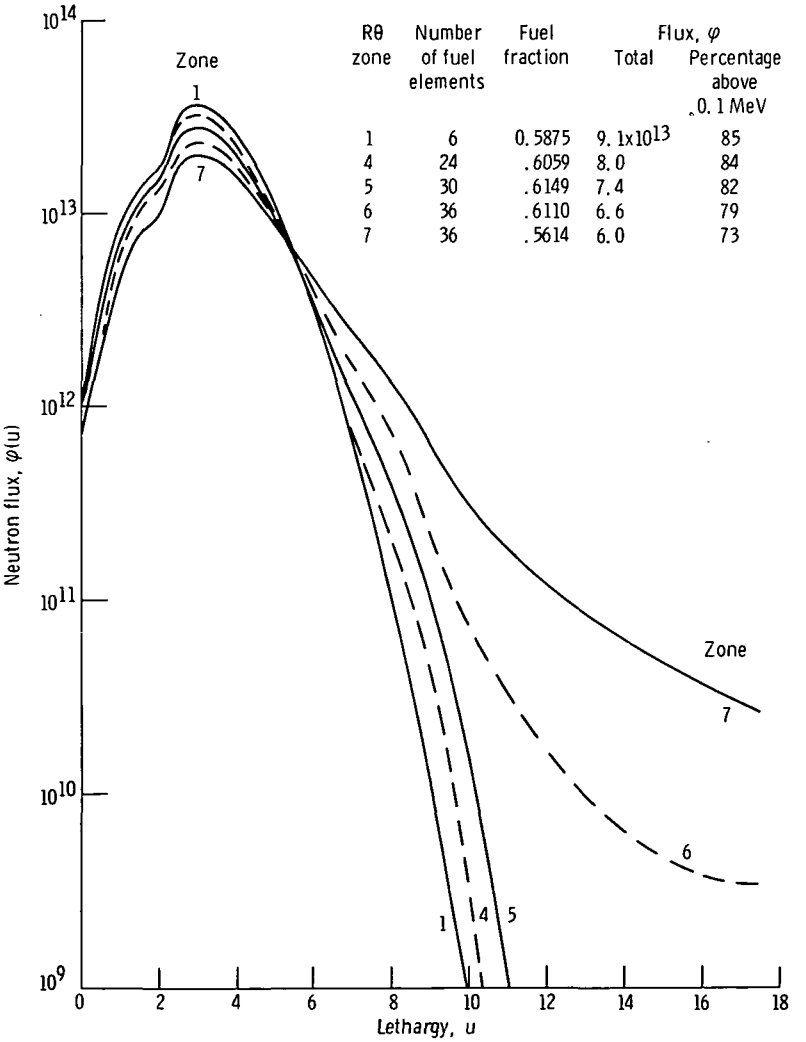

Figure 7. - Zone spectra from Rθ calculations. Power normalized flux, 1.575 thermal megawatts.

zones 2 and 3 are very similar to and would plot between zones 1 and 4 in figure 7. Note the increasingly softener spectra as one proceeds from the center of the reactor toward the reflector. As indicated, 85 percent of the flux is above 0.1 MeV for zone 1 but 73 percent for zone 7. Thus, with this amount of variation in spectral shape, it is advisable that fuel-element testing be done in appropriate spectra. The core average spectrum alone would not be appropriate.

In particular, the spectrum variation is quite extreme for the fuel elements nearest the reflector. Figure 8 shows spectra at the edge of the pin nearest the reflector and at the center of that same pin. This much softer spectrum at the pin edge is responsible for the large power peak observed at that point (fig. 6). The entire outer row of fuel elements would exhibit similar power peaking.

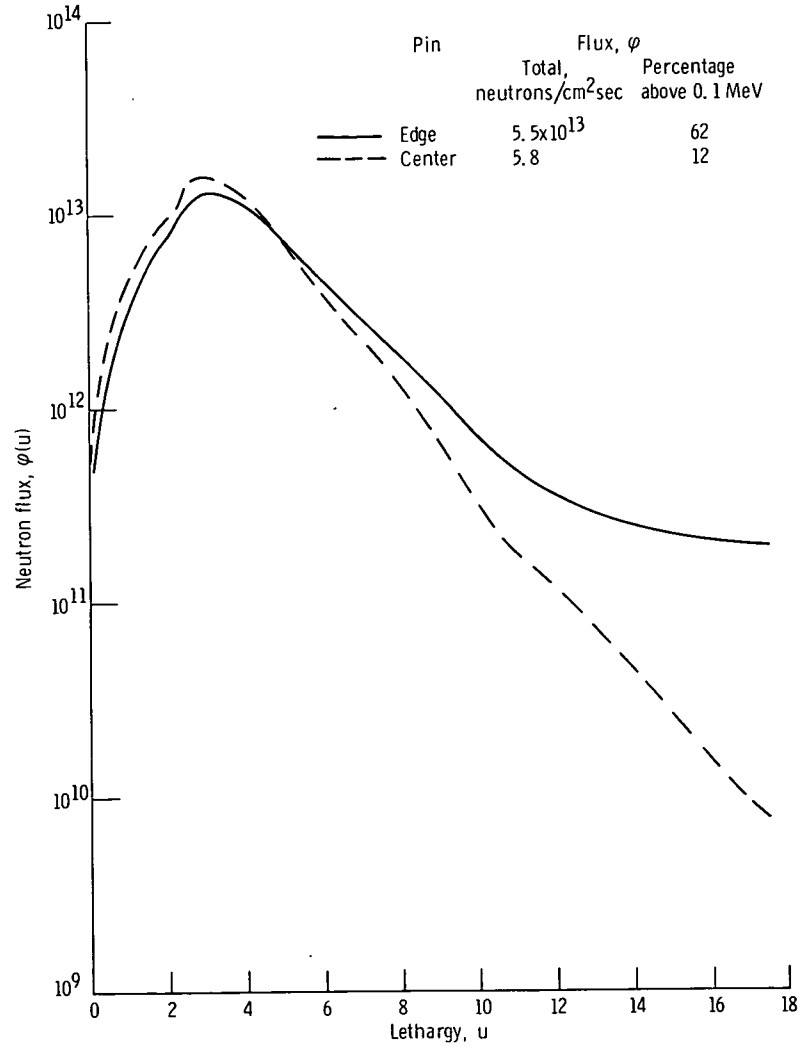

Figure 8. - Spectra at center and edge of RΘ zone 7 pin nearest pressure vessel. Power normalized flux, 1.575 thermal megawatts.

CONCLUDING REMARKS

An analysis procedure for reactors with radial and axial fuel zoning has been presented. The method can adequately treat the mixed spectrum problem encountered in the thermionic reactor in which a basically fast core is surrounded by a moderating reflector. Detailed power density distributions are presented. This type of configuration characteristically has a pronounced power spike at the core-reflector interface unless the peak is purposely suppressed by removal of the lower energy neutrons returning from the reflector. This could be done with an absorptive pressure vessel such as a tantalum based alloy instead of a niobium based alloy. The resultant decrease in reactivity and reactivity control would have to be evaluated and compensated for. With the demonstrated variation in spectra it is concluded that fuel-element proof-testing should be done in an appropriate neutron spectrum, or several spectra, that adequately represent the environment that would be encountered in the thermionic reactor. Use of a core average spectrum would clearly not be satisfactory.

Lewis Research Center,
National Aeronautics and Space Administration,
Cleveland, Ohio, May 16, 1973,
503-25.

APPENDIX - TREATMENT OF TRANSVERSE LEAKAGE IN DOT Rθ CALCULATIONS

For reactors in which the core spectrum and reflector spectrum do not differ greatly, the treatment of the transverse leakage by use of an effective buckling height has generally been adequate. The effective buckling height may be obtained by running one-dimensional radial calculations using the formula

$$\frac{1}{3\sum_{\text{trg}}^{i}}\left(\frac{\pi}{H_{\text{eff}}+\frac{1.42}{\sum_{\text{trg}}^{i}}}\right)^2$$

for transverse leakage cross section and varying the effective height H_{eff} until the multiplication factor agrees with that of an RZ calculation for the same reactor. The expression $\sum_{\text{trg}}^{i}$ is the transport cross section for energy group g and region i.

For fast-spectrum reactors with moderating reflectors in which a significant number of low-energy neutrons reenter the core from the reflector, a different approach to treating the transverse leakage in Rθ calculations is indicated.

Briefly, the RZ calculation is set up with zones such that, using the balance tables, the quantity

$$\sum_{L} \equiv \int_i \vec{J} \cdot d\vec{s} \Big/ \int_i \varphi \, dv$$

can be calculated. The surface integral of the neutron current J for zone i is performed over the surface s, which is normal to the transverse direction. This is divided by the integral of the neutron flux φ over the volume of the zone v. The leakage cross section $\sum_{\text{Lg}}^{i}$ for each group g and zone i is then used in the Rθ calculation. The Lewis DOT version has been modified to accept the $\sum_{L}$'s directly without having to incorporate them into the cross-section tables.

To illustrate the difference in results using these two methods, the TRCE B1 and B2 configurations were calculated.

Applicability to the Thermionic Reactor

No consistent H_{eff} method is currently available for treating the zoned reactor. The comparison for the TRCE configurations is, therefore, important as a check between methods and against experiment.

Description of the Calculations

A 22-energy-group structure was used with the S_4P_1 transport calculations. The GAF/GAR/GATHER programs were used to obtain the 13 fast-group - 7 thermal-group cross sections, except for stainless steel, which came from GAM. The element identification numbers are given in table II.

TABLE II. - FAST GROUP CROSS SECTIONS

Material	Identification number
Be	1007
O	1134
Al	1135
Cr	[a]24.0
Fe	[a]26.0
Ni	[a]28.0
W	1060 - 1063
U^{235}	1157
U^{238}	1158

[a]GAM.

RZ calculations, performed for each TRCE, were used to calculate the $\sum_L$ and were used to establish effective buckling heights. The $\sum_L$ and buckling heights were subsequently used in Rθ calculations. Regionwise $\sum_L$'s were used, that is, for the core, buffer, and reflector. A 67-by-12-Rθ mesh was used with fine mesh near the core periphery to compute fission density values.

Results and Discussion

Table III shows the effective buckling height for the various TRCE configurations. Table IV lists the multiplication factors obtained from the RZ calculations, from the Rθ

TABLE III. - EFFECTIVE BUCKLING HEIGHT

TRCE configuration	Effective height, H_{eff}, cm
B1	64.85
B1/2.93 BeO[a]	65.32
B1/2.85 BeO[a]	65.40
B2	64.27

[a]From ref. 1.

TABLE IV. - MULTIPLICATION FACTORS

TRCE configuration	Measured	RZ calculations	Rθ calculations	
			H_{eff}	$\sum_L$
B1	1.001030	1.0094	1.0398	1.0035
B1/2.93 BeO[a]	[b]1.02119	1.0319	1.0634	1.0255
B1/2.85 BeO[a]	--------	1.0279	1.0592	1.0206
B2	1.001294	1.0026	1.0292	.9940

[a]From ref. 1.

[b]From measured worth of 1/6 BeO reflector: 1.00103 + 6(0.00336).

calculations using buckling heights, and from the Rθ calculations using leakage cross sections, $\sum_L$. The Rθ calculations using $\sum_L$ are clearly in better agreement with experiment (and close to the RZ values) than the Rθ calculation using H_{eff}.

Table V shows fission densities, normalized to the core center, for the various TRCE configurations. The values calculated using $\sum_L$ are in closer agreement with the measurements than the values calculated using H_{eff}, except for the B2 core at the center of the flat (1.67) which is about 10 percent higher.

Figure 9 shows the leakage rate per source neutron from the Rθ calculations for the core of the B1 reactor by the H_{eff} method and by the $\sum_L$ method. The total leakage is also shown in the figure. The $\sum_L$ method indicates more high-energy leakage out of the core with significant return at lower energy. The H_{eff} method does not account for the low-energy return at all. Figure 10 shows the leakage rates for the B1 reflector. The two methods are in closer agreement for the reflector.

TABLE V. - CENTER-NORMALIZED PEAK FISSION DENSITIES FROM $R\theta$ CALCULATIONS

TRCE configuration	Fission density		
	[a]Measured	Calculation	
		H_{eff}	Σ_L
	Center or flat		
B1	3.751	3.18	3.45
B1/2.93 BeO[b]	2.836	2.55	2.83
B1/2.85 BeO[b]	-----	2.47	2.72
B2	1.506	1.48	1.67
	Apex or flat		
B1	-----	4.80	5.27
B1/2.53 BeO[b]	-----	3.70	4.13
B1/2.85 BeO[b]	-----	3.53	3.94
B2	2.52	1.83	2.11

[a] ±5 Percent experimental uncertainty.
[b] From ref. 1.

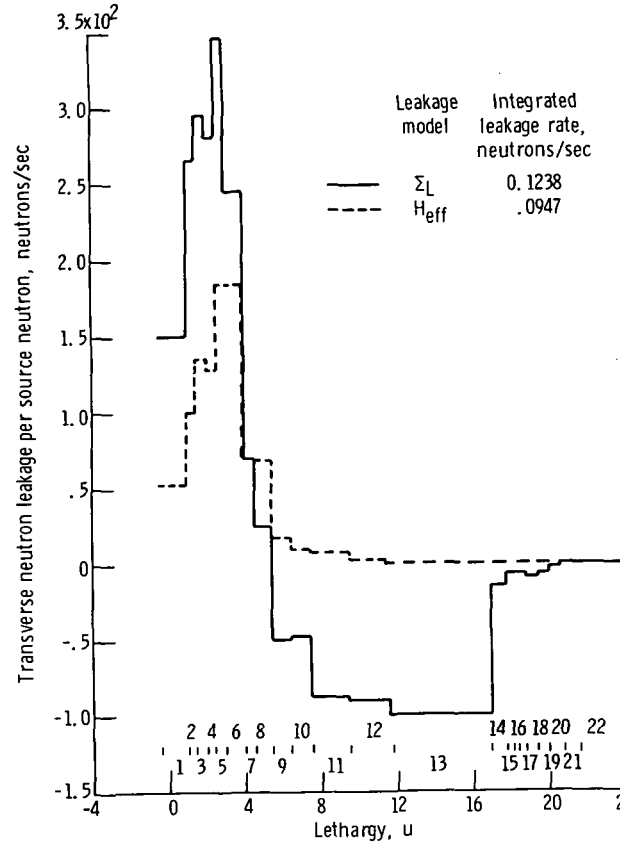

Figure 9. - Transverse leakage in B1 core.

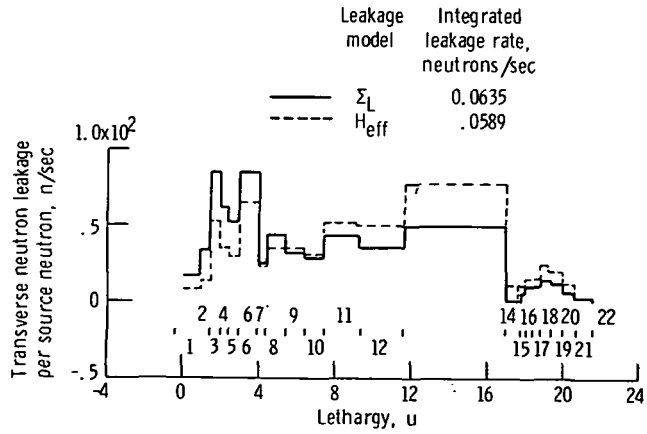

Figure 10. - Transverse leakage in B1 radial reflector.

On the basis of the limited comparisons done for the TRCE configurations, it appears that the $\sum_L$ method is clearly better than the H_{eff} method in calculating multiplication factors (table IV). For the fission density values at the core edge, a similar, but not so definitive, trend is also noted. The $\sum_L$ method is the most satisfactory method available for calculating the zoned thermionic reactor.

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